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USAEC - AECL COOPERATIVE PROGRAM

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E. I. du Pont de Nemours and Co.
Savannah River Laboratory
Aiken, South Carolina

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SECTION I

PHYSICS EXPERIMENTS WITH FUEL ASSEMBLIES SIMULATING BURNED-UP FUEL

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INTRODUCTION

Experiments are being performed in the Process Development Pile (PDP) and the Subcritical Experiment (SE) at the Savannah River Laboratory (SRL) to investigate the physics behavior of burned-up fuel in the CANDU and similar heavy-water power reactors. These experiments use specially fabricated fuel assemblies containing plutonium and uranium in approximately the isotopic compositions expected for fuel irradiated to 5000 MWD/ton. Separate sets of fuel assemblies also vary the total plutonium content and the isotopic fraction of ^{240}Pu.

SUMMARY

Analysis of the data from the FY-1966 PDP and SE experiments is in its final stages. However, some additional calibration measurements on fuel assays and foil perturbations have proven to be necessary in completing the SE studies. Chemical and isotopic analyses of the fuel have been completed at Savannah River and a composite sample of type "B" fuel is being sent to Chalk River for an independent test. In preparation for the FY-1967 experimental program, hot coolant loops are being installed in the SE.

DISCUSSION

The PDP substitution buckling measurements are being analyzed by the HERESY source-sink method. The necessary parameters for the driver lattices have now been determined. These include the thermal neutron absorption properties of the poison rods at the outer boundary (values chosen to match measured radial flux profiles) and ν, or η, its equivalent, which is chosen to fit the measured critical water height. HERESY parameters for the test fuel have been computed by the multigroup HAMMER code using a ring model for the rod clusters and for the chemical and isotopic fuel compositions (the SRL measurements) listed in DPST-66-83-5.

Initial results of the PDP data analyses suggest that, with respect to internal consistency, the Persson perturbation method of successive substitution analysis and the HERESY source-sink method have approximately equal precision and that the final test lattice bucklings derived from the two methods should be nearly identical.

The completion of the SE experiments, notably the foil assays by neutron activation, is being somewhat delayed by foil counting capabilities. Despite round-the-clock utilization of the two counting systems available for this work, a large backlog still exists.

The discrepancies between NFS, ORNL, and SRL in the determination of the Pu content of the various fuel rods have still not been resolved. However, pending an independent analysis of the Type B fuel (nominal 0.00266 Pu/U atom ratio) by Chalk River, the SRL results, which show good internal consistency between several different techniques, are being used.

The SRL FY-1967 program with the burned-up fuel is to consist primarily of fuel and coolant temperature coefficient measurements as described in Table I-I. In preparation for these experiments it is necessary to install special high temperature loops in the SE. Design of these loops is essentially complete. The loops are intended primarily for organic coolant, although water can also be used at temperatures up to the 300 psi pressure limit. They consist of separate heating and cooling units connecting to a manifold in the SE room. Flexible tubing will be used to connect a full SE load of test fuel to the manifold.

TABLE I-I

TEMPERATURE COEFFICIENT MEASUREMENTS - BURNED-UP FUEL MOCKUPS

Lattice Pitch, Δ, in.	Cluster Size	Pu Content Total	Pu Content % 240	Test Coolant	Fuel Temp., °C	Measurements
9.33	31	0.266	6	D_2O	20-80	B^2, ρ^{28}, Lu/Cu
9.33	31	0.266	6	HB-40	20-350	B^2, ρ^{28}, δ^{28}, δ^{25}, Lu/Cu, Pu/U
9.33	31	0.266	6	Air	20-350	B^2, ρ^{28}
9.33	31	0.309	19	HB-40	20-350	B^2
9.33	31	0.372	6	HB-40	20-350	B^2
9.33	31	Natural Uranium		HB-40	20-350	B^2
12.12	31	0.266	6	D_2O	20-80	B^2, ρ^{28}, Lu/Cu
12.12	31	0.266	6	HB-40	20-350	B^2, ρ^{28}, δ^{28}, δ^{25}, Lu/Cu, Pu/U
12.12	31	0.266	6	Air	20-350	B^2, ρ^{28}

All measurements to be made in the SE with D_2O moderator at 20°C and the rod pitch in the clusters at 0.607".

ACTIVATION MEASUREMENTS:

- ^{176}Lu/Cu = Activation Ratio Measurement of Neutron Temperature
- ^{239}Pu/^{235}U = Activation Ratio Measurement of Neutron Temperature
- $\rho 28$ = Epithermal-to-Thermal Capture of Ratio in ^{238}U
- $\delta 28$ = Total ^{238}U Fissions/Total ^{235}U Fissions
- $\delta 25$ = Epithermal-to-Thermal Fission Ratio in ^{235}U
- C* = ^{238}U Captures/^{235}U Fissions
- Mn, Au, In = Activation Flux Traverses

SECTION II

AECL IN-CORE FLUX MONITORS

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INTRODUCTION

An irradiation test of in-core flux monitors is being made in one of the Savannah River Plant reactors to determine the life characteristics of a selection of flux detectors and of the mineral insulation used in their construction. Self-powered flux detectors are relatively new; therefore, confidence in their use hinges to a great extent on proven performance at large integrated exposures. The chief points of interest are 1) integrity of the conductors and sheath during life, 2) life of insulation, and 3) sensitivity at SRP (vis-a-vis Chalk River) will shorten the irradiation time for a given exposure and should also show whether or not any new high intensity effects appear.

SUMMARY

Fabrication and installation of the detector rod in the reactor has been completed and testing is in progress.

DISCUSSION

Irradiation testing of the AECL neutron detector rod (described in DPST-66-33-5) in the Curium II reactor is continuing.

An analysis of the data since the beginning of the test indicates that either the Zircaloy detector or the cable has failed. The current output from the Zircaloy detector has been erratic throughout the test period. The detector produces either negative or positive currents during reactor operation; although it normally produces a negative output (approx. 3×10^{-10} amperes) during reactor shutdown. The data for the Zircaloy detector will be collected for the duration of the test.

The data for all detectors and cables are being collected and analyzed. They will be reported in a separate topical document at the conclusion of the tests.

SECTION III

SIEVE TRAY TEST

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INTRODUCTION

The maximum fluid handling capacity of sieve trays enters into comparisons of future power costs for various types of nuclear reactors through its effect on the projected cost of heavy water. In 1963 an expenditure of $230,000 was authorized to modify equipment and evaluate the capacity of sieve trays in one of the idle GS units at the Savannah River Plant.

To reduce construction and operating costs, the test unit was interconnected with and limited by the pressure in the existing GS Plant. While these modifications were in progress, GS Plant pressure was reduced from 275 psig to 225 psig (at the proposed point of interconnection) until the extent of external corrosion of carbon steel equipment could be determined.

SUMMARY

Preliminary tests at 225 psig, initially with segmental downcomers and finally with downpipes, were completed in July 1964. The tests were conducted by circulating H_2S countercurrent to water through seven sieve trays in a tower 6-1/2 feet in diameter. Liquid entrained in the gas from the third to the fourth tray and differential pressures across the various sections were measured, while holding the liquid-to-gas molar flow ratio (L/G) at the optimum required in an operating plant. At 225 psig, operation with an L/G of 0.50 at 34°C (GS cold tower conditions) was simultaneously limited by tower flooding and blower capacity to a maximum F-factor* of about 1.65. Segmental downcomers and downpipes performed alike. These results, along with a complete description of the test facility, are contained in the Monthly Progress Reports for February, March, and July 1964 (DPST-64-83-2, -3, and -7) and in USAEC Report DP-1025 (June 1966) Performance of Sieve Trays Under GS Heavy Water Process Conditions, Part I.

PROGRESS

GS Plant pressure, at the point of test facility interconnection, was increased to 275 psig early this year and sieve tray capacity tests, with downpipes, were resumed on July 13, 1966. During initial tests

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* All F-factors in this report are based on 31.9 sq ft of free area in the 6'6" diameter tower. $F = \mu\sqrt{\rho}$; μ = gas velocity, ft/sec; ρ = gas density, lb/ft^3.

at this pressure, stable operation at GS cold tower conditions was limited by tower flooding to F-factors of 1.3 to 1.5 without silicone antifoam addition and 1.4 to 1.6 with 1 ppm of "GE-60"* silicone in the feedwater. The flooding was apparently caused by poor feedwater quality or perhaps particulate matter (foam nuclei) in the equipment after a two-year shutdown. In a subsequent run, stable operation was achieved, with 1 ppm silicone, at an F-factor of 1.81 and flooding occurred at blower capacity, F-factor 1.87. Tray-to-tray entrainment at the flooded condition was about 3 mols liquid per 100 mols gas. Pressure drop data for this run are shown in Figure III-1. This capacity has been verified with a duplicate run in which stable operation was maintained up to an F-factor of 1.75 and flooding occurred at F-factor 1.80. Also during a run with gas flow constant at F-factor 1.80 and increasing liquid flow, stable operation was maintained up to a liquid flow of 148 gpm (L/G = 0.47); flooding began at a liquid flow of 152 gpm (L/G = 0.50). Complete data for these runs and additional tests now in progress will be presented next month.

TEST PROGRAM

Tests at constant liquid flow and increasing gas flow are now in progress. An additional run at constant F-factor and increasing liquid flow will be made. Work will be completed and the test unit shut down and depressured in early August.

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*General Electric Co., Schenectady, New York.

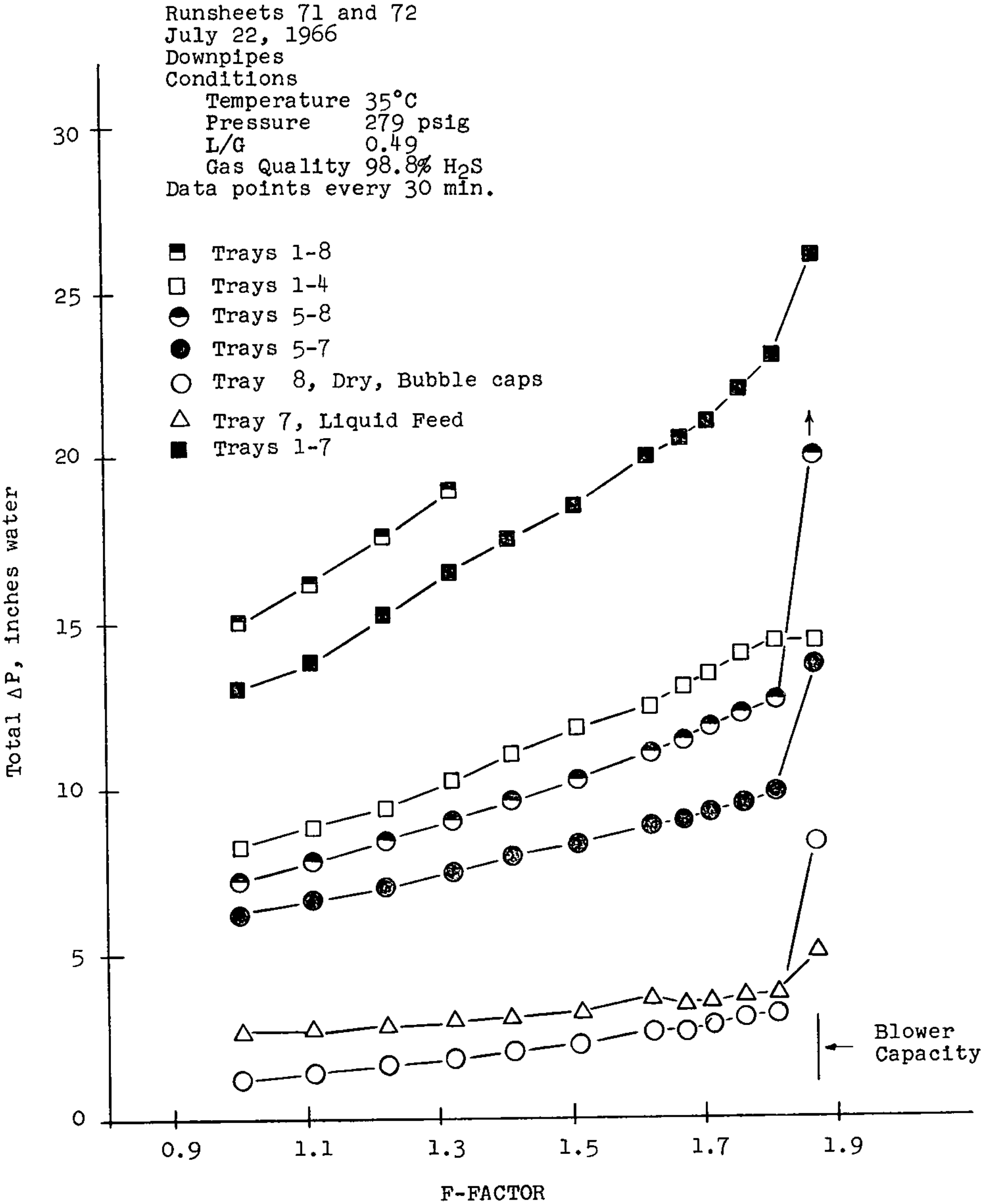

FIG. III-1 PERFORMANCE OF SIEVE TRAYS UNDER GS COLD TOWER CONDITIONS

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